MONTE CARLO AND DETERMINISTIC EVALUATION OF PREESSURE TUBE CREEP IN A CANDU LATTICE

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Abstract

In the CANDU reactor, the coolant channel diameter increases due to the pressure tube creep resulting from neutron irradiation damage. For two diametric creep (2.5% and 5%) cases in the standard 37-pin CANDU lattice, the pressure tube creep was evaluated from the neutronic point of view with the continuous-energy Monte Carlo depletion code McCARD and the collision probability neutron transport code HELIOS. Impact of the crept pressure tube is evaluated in terms of the lattice reactivity, void reactivity, pin-wise power, atom density of plutonium isotopes, and fuel temperature coefficient. The results show that both McCARD and HELIOS codes provide consistent results and the coolant void reactivity increases noticeably with the crept geometry.

1. Introduction

In the CANDU reactor, the coolant channel diameter increases due to the pressure tube creep resulting from neutron irradiation damage. Along with the creep, the pressure tube also undergoes a slight sagging phenomenon. It is well known that a crept pressure tube leads to an increased by-pass flow in the coolant channel and the operational margin of the reactor is reduced.

In a crept geometry, the coolant volume is increased and it is naturally speculated that the increased coolant volume can increase the positive coolant void reactivity further, deteriorating the safety of the CANDU system. Previously, Ilas et al[1] analysed the neutronic impact of the pressure tube creep on the physics parameters and concluded that its impact is not insignificant.

In this work, the pressure tube creep problem is revisited and its neutronic impact is evaluated with a continuous-energy Monte Carlo McCARD[2] and the HELIOS code[3]. McCARD has a built-in depletion module and the depletion analysis can be done in stand-alone manner. Impact of the crept pressure tube is evaluated in terms of the lattice reactivity, void reactivity, pin-wise power, atom density of plutonium isotopes, and fuel temperature coefficient. For the nominal model of CANDU lattice, the result of WIMS-AECL [4] calculation is also added for a comparison purpose.

2. CANDU lattice model and analysis methods

In this work, the standard 37-pin CANDU lattice is used and two values of diametric creep (2.5% and 5% of nominal) are considered. Figure 1 shows HELIOS-1.8 models for the reference and 5% creep case. Natural uranium is used as the fuel and, in normal operating conditions, temperatures of fuel, coolant, and moderator are 960 K, 561 K and 342 K, respectively. The D_2O purities of the coolant and moderator are 99.10 wt% and 99.833 atom%, respectively, and the corresponding densities of the coolant and moderator are 0.807859 g/cm^3 and 1.085089 g/cm^3, respectively.

In the reference case, reflective boundary conditions are imposed on all the boundaries, while a periodic boundary condition is used on the top and bottom surfaces due to the asymmetry of the

perturbed lattice. The HELIOS calculations are done with an adjusted 190-group cross section library which is based on ENDF/B-VI. A 3.4% reduction of the capture cross-section of the U-238 isotope is applied for the adjusted library in the resolved resonance range. The pin-wise depletion calculations are utilized. Regarding the mesh coupling in the HELIOS calculations, the exact coupling option is used in both nominal and perturbed models.

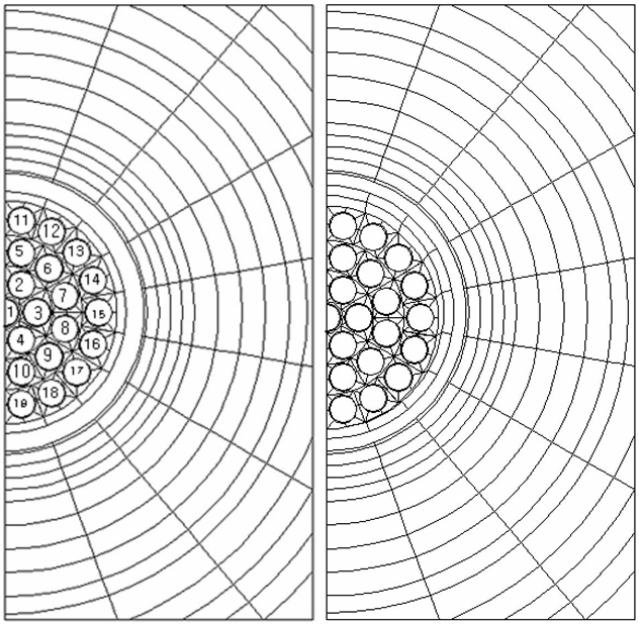

Figure 1 Nominal and perturbed lattice models.

In parallel with HELIOS calculations, the CANDU lattice crept problems are also analysed with the McCARD code for a comparison. McCARD is a Monte Carlo neutron transport code and basically designed for the depletion analysis for the nuclear power reactors. The code can handle arbitrary system geometry by dividing it into three-dimensional unit cells that are defined using surfaces such as planes, cylinders and sphere. Because McCARD code is basically designed for the depletion analysis of nuclear power reactors, McCARD has built-in subroutines to solve the depletion equation while other Monte Carlo depletion codes are usually coupled with independent depletion code. The code uses continuous energy nuclear data cross section library generated by the ACER module of NJOY. In this study, the continuous cross section library is based on ENDF/B-VII.

For the nominal model of CANDU lattice, the result of WIMS-AECL calculation is also added for a comparison purpose. In this study, WIMS-AECL Release 2-5d was used with 89 energy group cross section library based on ENDF/B-VI. This version is the so-called Industry Standard Toolset and is used for design, safety analysis and operational calculations of the physics of natural-uranium-fuelled CANDU reactors. However, because the WIMS-AECL Release 2-5d is not able to model the eccentric location of fuel bundles in the CANDU lattice, the WIMS-AECL calculations were only performed for the nominal CANDU lattice to compare with the HELIOS and McCARD codes.

3. Analysis Results

In Figure 2, the infinite multiplication factors obtained by HELIOS, McCARD and WIMS-AECL codes are compared for the nominal CANDU lattice. In the McCARD depletion calculations, over

200 nuclides are considered including all actinides and each fuel element is independently depleted. The statistical uncertainty of the McCARD results is less than 0.09 mk for the k-infinity values.

The result of HELIOS calculation shows a good agreement with that of WIMS-AECL calculation. The McCARD code underestimates the infinite multiplication factor compared to the HELIOS and WIMS-AECL codes. It is because the cross section library of HELIOS code is adjusted for then capture cross section of U-238, while the no adjustment is applied to the McCARD cross section library. Compared with the results of WIMS-AECL, the HELIOS and McCARD codes overestimate by 2.9 mk and 1.4 mk, respectively, at the fresh fuel state (0 MWd/tU). When the fuel burnup is above 500 MWd/tU, the McCARD code underestimates reactivity by ~3.5 mk.

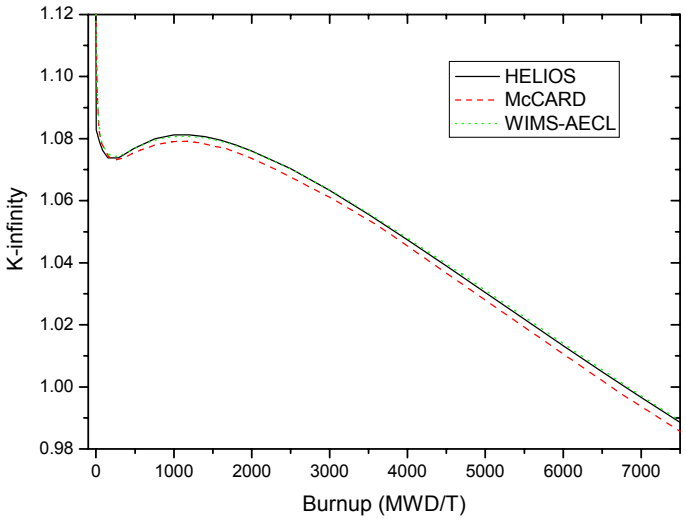

Figure 2 Comparison of the lattice reactivity vs. burnup.

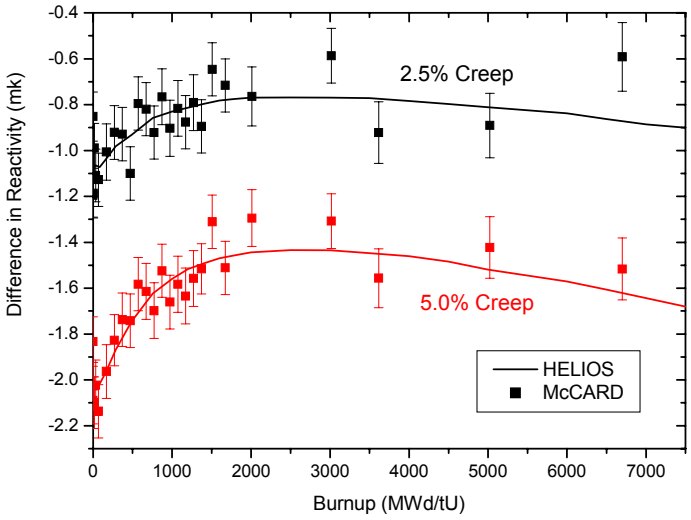

Figure 3 Comparison of the lattice reactivity vs. burnup.

As shown in Figure 3, in the cases of a crept pressure tube, the lattice reactivity is decreased noticeably. The reactivity reduction ranges -1.1 ~ -0.8 mk for the 2.5% creep and -2.0 ~ -1.5 mk for the 5% creep case. It is expected that the discharge burnup will be reduced by a few percents in the case of 5% creep. Generally, the results of McCARD calculation agree well with those of the

HELIOS calculation in terms of the reactivity difference induced by the pressure tube creep. The reduced reactivity is mainly ascribed to the increased coolant volume in the crept models.

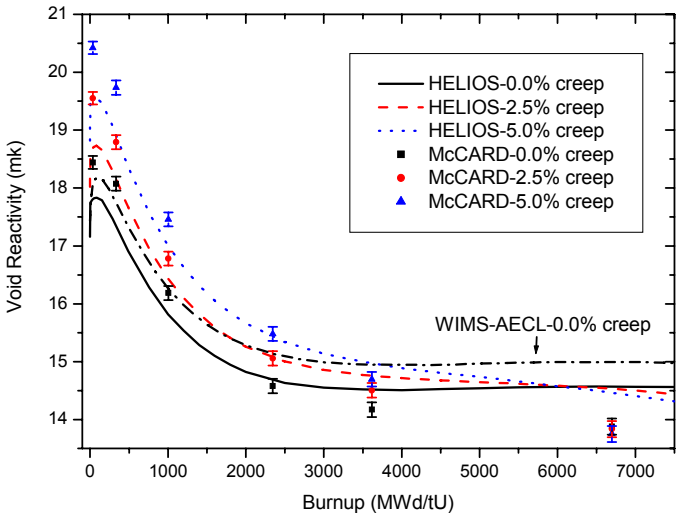

Figure 4 Impact on the coolant void reactivity.

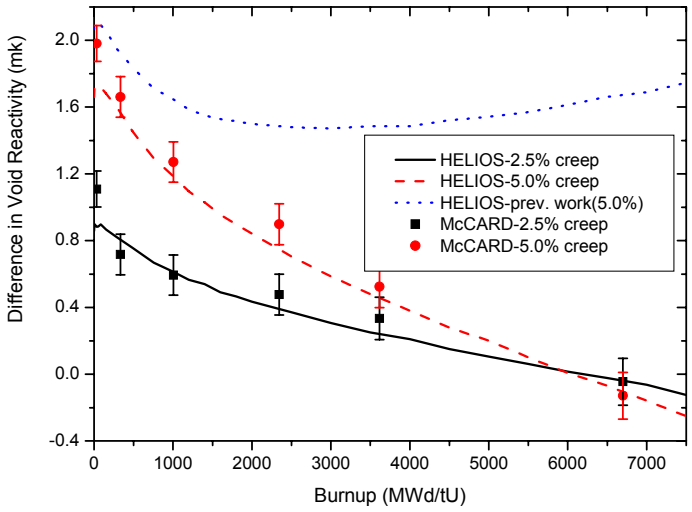

Figure 5 Difference in coolant void reactivity.

The coolant void reactivity was evaluated for the three lattice models and the results are shown in Figure 4. For the nominal cases, the result of the WIMS-AECL calculation is also compared. The result of HELIOS underestimates by 0.5 mk and the discrepancy of the void reactivity is almost same for a whole burnup states. At the same time, the result indicates that the void reactivity increases with the creep. Compared to the result of McCARD code, HELIOS code underestimates at

the low fuel burnup, and overestimates at high burnup, and the maximum difference between two codes is ~0.9 mk.

Figure 5 shows the difference in void reactivity between reference and crept lattices. Figure 5 also compares the current results with a previous work [1] for the 5% creep case. The results indicate that the void reactivity increases with the creep and the difference is rather significant (~2 mk at zero burnup for 5% creep) at low burnup and decreases with the fuel burnup. At mid-burnup near 3000 MWD/T, the differences are ~0.4 mk and ~1 mk for 2.5% and 5% creep, respectively. In addition, Figure 5 indicates that the results from MCCARD and HELIOS agree well. However, a clear difference is observed between the current and previous works. In the previous result, which was obtained with HELIOS-1.7, the void reactivity difference is rather big over the whole burnup range. The reason for the difference is not identified at the moment and it is under investigation.

In Figure 6, the fuel temperature coefficient (FTC) is compared for the three cases. It is observed that the impact of the pressure tube creep on the FTC is extremely small or negligibly small when the fuel burnup is relatively low. However, it is worthwhile to note that the FTC becomes slightly less negative or more positive when the burnup is above about 2,500 MWD/tU. The small change cannot be neglected since the FTC of the CANDU lattice is already close to zero in the vicinity of the normal operating conditions and a small change in FTC can lead to a change in the sign of the FTC, from a negative to a positive FTC.

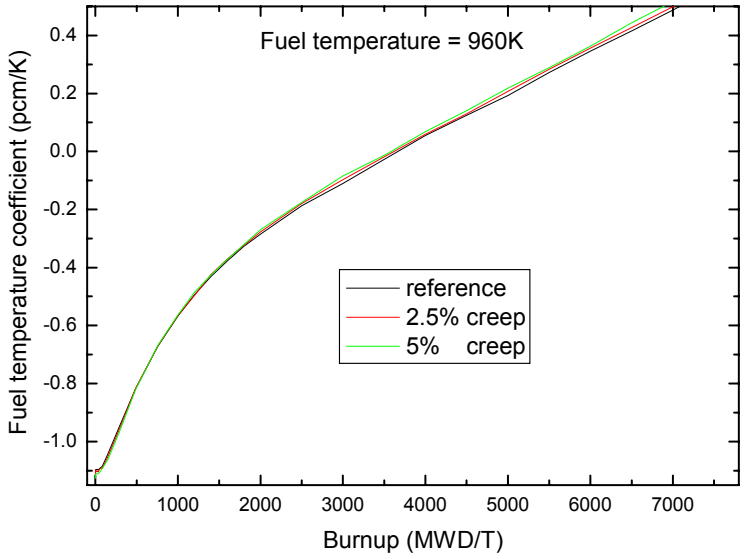

Figure 6 Impact on the fuel temperature coefficient.

Figure 7 shows that the pin-power distribution is also affected noticeably by the pressure tube creep. There is an increase in the pin power in downward direction. The bottom-most pin (#19) undergoes the biggest increase, while the top-most one (#11) shows the biggest decrease. The difference decreases with the fuel burnup. For the three cases, the maximum normalized pin-powers over the whole burnup range are 1.136, 1.140, and 1.143, respectively. Figure 7 also indicates that both McCARD and HELIOS codes provide comparable lattice power distributions over the whole burnup range.

Plutonium isotopes play an important role in the safety-related parameters of CANDU. Figure 8 shows the impact of the 5% creep on the actinide (U and Pu) atom density of the lattice. Overall, difference of the Pu atom density decreases with burnup, except that Pu-239 atom density is lower at 5% creep over the high burnup range.

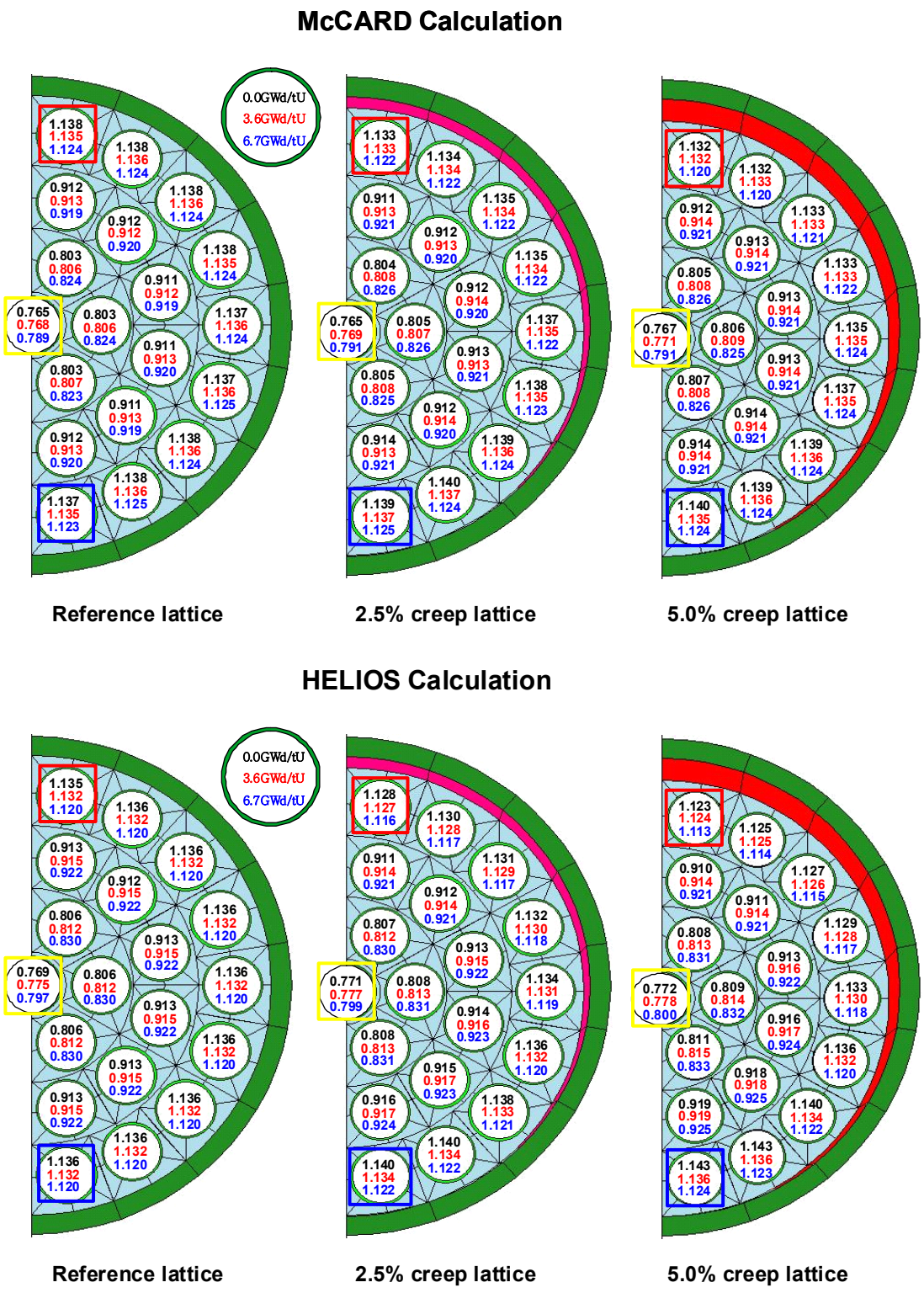

Figure 7 Impact on the pin-power distribution.

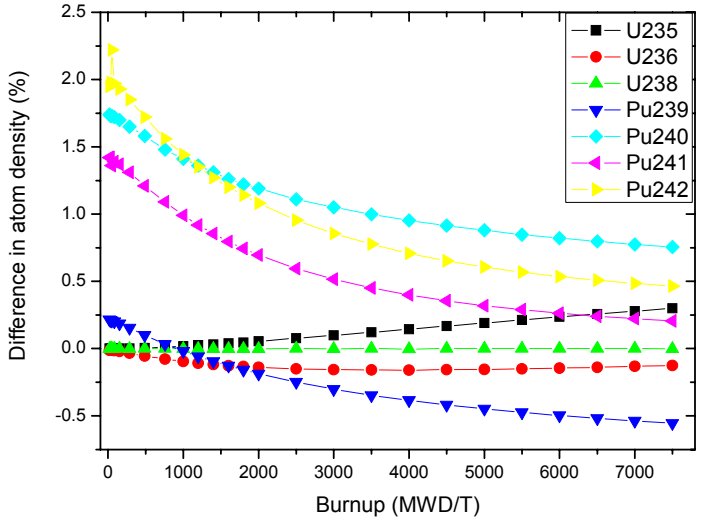

Figure 8 Impact on the actinide density (5% creep, HELIOS calculation).

4. Conclusion

In the standard CANDU lattice, the impact of the pressure tube creep on the physics parameters is noticeable. In particular, the coolant void reactivity increases noticeably with the crept geometry. The pressure tube creep results in a slightly reduced lattice reactivity and a slightly higher power peaking in a fuel bundle. The fuel temperature coefficient becomes slightly less negative or more positive when the fuel burnup is above about 2,500 MWD/tU with the pressure tube creep. All the safety-related physics parameters are affected by the pressure tube creep. Therefore, the irradiation damage of the pressure tube should be duly considered in the nuclear design and safety analysis of a CANDU core.

5. References

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[2] H. J. Shim and C. H. Kim, "MCCARD - Monte Carlo Code for Advanced Reactor Design and Analysis, User's Manual Version 1.0," Nuclear Design and Analysis Laboratory, Seoul National University.

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[4] S. R. Douglas, "WIMS-AECL Release 2-5d Users Manual," RC-1176/COG-94-52(Rev.4)/FFC-RRP-299, Chalk River Laboratory, July 2000.